ACCESSION #: 9609050107

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Limerick Generating Station, Unit 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000352

TITLE: Core Thermal Power Exceeded Licensed Power Limit During

Power Transient Caused By Defective EHC System Component

& Reactor Scram Resulting From Various Feedwater Heater

Isolations

EVENT DATE: 07/25/96 LER #: 96-016-0 REPORT DATE: 08/26/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

OTHER

LICENSEE CONTACT FOR THIS LER:

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Assessment, LGS

COMPONENT FAILURE DESCRIPTION:

CAUSE: X SYSTEM: JI COMPONENT: CNV MANUFACTURER: G080

REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 07/25/96, at 0224 hours, Unit 1 experienced a power transient due to a sudden momentary opening of main turbine by-pass valves and partial closure of the control valves. Indicated APRM neutron flux increased to 113.75% over a 10 second period.

Analysis indicates that actual peak heat flux was 107.5%. This event was bounded by the licensing basis of the core. In addition, at 0242 hours, an automatic reactor scram occurred on high neutron flux caused by a loss of feedwater heating due to the isolation of various feedwater heaters following recovery from the power excursion. The plant responded as designed to the high neutron flux signals. The cause of the power excursion was a malfunction of the primary speed frequency/voltage (F/V) converter in the main turbine electrohydraulic control system. The defective primary speed F/V converter was replaced. The cause of the reactor scram was less than adequate procedural guidance regarding power reduction in response to reactivity insertion resulting from the loss of feedwater heating. The emergency operating procedure was revised to provide an appropriate target power level on the loss of a feedwater heater string.

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Unit Conditions Prior to the Event

Unit 1 was in Operational Condition (OPCON) 1 (Power Operation) at 100 percent power level. There were no systems, structures, or components out of service which contributed to this event.

Description of the Event

At 0224 hours on July 25, 1996, the "Bypass Valve Open" and "Rod Out Block" alarms annunciated in the Main Control Room (MCR). The Unit 1 Reactor Operator (RO) looked at the main turbine bypass valve indicators and noticed that three (3) bypass valves (EIIS:PCV) were open. All nine (9) bypass valves then cycled full open and then full closed. This was immediately followed by an "APRM Upscale" alarm and a B channel half scram signal due to the 1F Average Power Range Monitor (APRM) in the "HI-HI" trip status.

Recognizing that an unexpected change in reactivity had occurred, the MCR Shift Supervisor (SSV) entered Operational Transient (OT) procedure OT-104, "Unexpected/Unexplained Reactivity Insertion." The initial command to the RO was to reduce reactor power in accordance with the Reactor Maneuvering Shutdown Instructions (RMSI). The Shift Manager (SM) and SSV subsequently concluded that reactor power should be reduced to 90 percent using reactor recirculation flow until the cause of the bypass valve perturbation could be determined. By 0226 hours, power reduction was in progress.

At 0228 hours, the feedwater side of the Unit 1 C low pressure feedwater heater string isolated on high level in the 2C feedwater heater (EIIS:HX). Each unit has three (3) strings of feedwater heating (i.e., A, B and C strings) with six (6) heaters in each string (labeled 1 through 6). Feedwater inlet temperature was confirmed to be within the operating region of the Core Power/Flow Log, and a decision was made to restore the isolated C feedwater heater string. While restoring feedwater flow through the C feedwater heater string, the extraction steam supplies to the 3B and 4B feedwater heaters isolated due to high level in the heaters. At 0232 hours, the C feedwater heater string was restored in accordance with the appropriate system procedure.

By 0230 hours, reactor power had been reduced to 90 percent. The MCR operators then attempted to determine the cause of the main turbine

bypass valve transient. The crew checked the turbine control valve position recorder and identified a spike in the total turbine control valve position. The SM, SSV and Assistant Control Room Supervisor (ACRS) immediately recognized that a main turbine electrohydraulic control (EHC) system (EIIS:JI) malfunction had occurred. The SM directed the crew to reduce power to less than 25 percent, within bypass valve capability, to protect the reactor from a potential turbine trip from an EHC system transient. Direction was given for the RO to manually scram the reactor if any additional perturbations were observed.

At 0232 hours, additional extraction steam supply isolations occurred on the 3C and 4C feedwater heaters due to high level. By this time, final feedwater temperature had dropped from 428 degrees F to 416 degrees F.

At 0238 hours, the RO was in the process of reviewing the RMSI to begin inserting control rods and reduce reactor power when the 5B and 5C feedwater heaters isolated. At 0240 and 0241 hours, the 6C and 6B feedwater heaters also isolated, respectively. With these feedwater heaters isolated, final feedwater temperature rapidly dropped to 336 degrees F, causing reactor power to increase. At this time, the RO observed reactor power to be approximately 95 percent and increasing.

Based on these additional perturbations, the RO proceeded to manually scram the reactor, concurrent with verbal direction from both the SM and SSV, when the reactor automatically scrammed at 0242 hours on high neutron flux. The SSV entered Transient Response Implementation Plan T-101, "RPV Control RC/Q, RC/L, RC/P," and performed all necessary steps.

Total feedwater flow was reduced to zero within one minute after the scram occurred. However, the cold feedwater injection prior to the scram combined with control rod drive injection caused reactor water level to rise to a maximum of +68 inches. At this time there was no feedwater injection occurring. Procedure OT-110, "Reactor High Level," was entered. There was no need to manually open main turbine bypass valves or safety relief valves because level was maintained below +70 inches. Subsequent followup actions were performed to place the plant in a stable condition.

A four hour notification was made to the NRC at 0433 hours on July 25, 1996, in accordance with the requirements of 10CFR50.72(b)(2)(ii) since this event involved an automatic RPS actuation. In addition, this notification satisfied the requirement for a 24 hour notification as

specified in the LGS Unit 1 Facility Operating License, Condition 2.F, since this event resulted in a non-compliance with License Condition 2.C.1. License Condition 2.C.1 provides authorization to operate the Unit 1 reactor at a maximum reactor core power level of 100 percent rated power. This report is submitted in accordance with the requirements of 10CFR50.73(a)(2)(iv), and in accordance with the requirements of License Condition 2.F, which requires a 30-day followup written report.

Analysis of the Event

The consequences of this event were minimal and there was no release of radioactive material to the environment as a result of this event.

The entire EHC excursion lasted approximately 25 seconds with indicated

APRM neutron flux greater than 100 percent for less than 10 seconds. A review of computer data showed peak neutron flux on the 1F APRM was 113.75 percent. An analysis of core thermal power was performed using the neutron flux response from the APRMs as input to a transient fuel bundle simulation model. The results of this analysis indicated that the peak core heat flux during the transient was approximately 107.5 percent of the licensed limit.

The Supplemental Reload Licensing Report for Limerick Generating Station (LGS), Unit 1, Reload 6, Cycle 7, dated January 1996, provides the results of the analyses of the most limiting transient events. The peak core heat flux (Q/A) for the most limiting events (with no systems out of service) ranges from 112 percent to 116 percent. These results determine the Operating Limit Minimum Critical Power Ratio (OLMCPR) for the cycle. The OLMCPR insures that the Safety Limit MCPR (SLMCPR) will not be violated in the event that a plant transient should occur. The peak core heat flux during this event was much less severe than that for the limiting events in the Reload Analysis. Therefore, the event was bounded by the licensing basis of the core and no Technical Specifications thermal safety limit violation occurred.

The plant responded as designed to the high neutron flux scram signal. All control rods properly inserted. The maximum reactor pressure observed during the EHC transient was approximately 1078 psig, which is below the scram setpoint of 1096 psig and well below the Technical

Specifications (TS) Safety Limit of 1325 psig as measured in the

reactor steam dome. The main steam relief valves would have lifted as necessary to maintain reactor pressure below the Safety Limit.

A review of the event and resultant actions by the shift operating crew was performed. This review concluded that the crew's response and operator actions were in accordance with the applicable procedures and were appropriate. For example, based on shift supervision direction, the RO had already armed the manual 'scram push buttons in the MCR and was in the process of manually scramming the reactor when the automatic scram occurred. The crew briefings and decision making during the event were in accordance with the current conservative decision making practices.

Cause of the Event

The cause of the initial reactor power transient was determined to be an intermittent failure of the primary speed F/V converter card (EIIS:CNV), GE Model No. 115D3332G3, in the EHC system's speed control circuit (EIIS: SC).

On July 22, 1996, the EHC system's speed control circuit had automatically transferred from the primary amplifier to the secondary (or backup) amplifier as evidenced by an "EHC Malfunction" alarm in the MCR. This transfer was discussed with General Electric (GE), and the decision was made to continue running on the secondary speed card while performing further investigation and planning for corrective actions. Subsequent troubleshooting on July 24, 1996, identified that the output of the primary F/V converter card in the speed control circuit was indicating lower than expected, i.e., a voltage that corresponded to a turbine speed of 1710RPM instead of 1800RPM.

At 0224 hours on July 25, 1996, the EHC system's Speed-Demand Error (SDE) instantly changed from the expected value of approximately 0.035VDC to +1.7VDC. A subsequent review of computer data indicated that the MCR "EHC Malfunction" alarm had cleared coincident with the onset of the event indicating that the primary speed amplifier had regained control.

A sudden increase in the output of the malfunctioning primary F/V converter resulted in the change in SDE which caused Control Valve Demand (CVD) for the main turbine control valves to decrease from 100 percent to approximately 45 percent, and Bypass Valves Demand (BPVD) for the main turbine bypass valves to increase to 100 percent. The SDE then began to ramp back to a value of

approximately 0.0VDC at a rate corresponding to 180RPM/minute. The main turbine control valves began to respond at their normal rates to the corresponding changes in CVD back to 100 percent, while the bypass valves tried to control reactor pressure in response to pressure regulator output (i.e., closing to 30 percent and reopening to 100 percent). EHC pressure control continued to function as expected following the event.

The cause of the reactor scram was less than adequate procedural guidance addressing reactivity insertion resulting from the loss of feedwater heating. Operations personnel entered procedure OT-104 in response to the initial unexplained reactivity insertion. This procedure instructs the operator to reduce power to avoid a scram, but provides no target power level. The additional guidance given in procedure OT-104 to address the reduction in feedwater temperature does not provide sufficient information to assure feedwater heating system stability relative to reactor power level. With the loss of one low pressure heater string, excessive flow through the remaining two heater strings imposed excessive heating load requirements which led to additional loss of feedwater heating and the subsequent reactor scram.

A contributing factor to the series of feedwater heater isolations was the failure of the 2C feedwater heater drain valve to maintain proper heater water level. Subsequent inspection of the valve identified a stem seal failure in the valve actuator resulting in the inability of the valve to function correctly.

Corrective Actions

The defective primary F/V converter was replaced and a calibration performed on the speed control circuit. No other deficiencies were identified. Prior to exceeding 25 percent reactor thermal power during startup, and following synchronizing the Unit 1 generator to the grid, additional speed control circuit tuning was performed and the system responded as expected.

A failure analysis of the defective F/V converter card is in progress to determine the cause of its malfunction. Any additional corrective actions will be evaluated based on the results of the failure analysis.

Because of the uniqueness of the EHC system components, this event is not applicable to other plant control systems.

Procedure OT-104 was revised on July 30, 1996, and specifies a target power level for reactor power reduction in response to reactivity insertion caused by the loss of feedwater heating due to the isolation of a low pressure feedwater heater string. This procedure change provides operators with an appropriate power level for two feedwater heater string operation and will assure feedwater heating system stability during similar plant transients.

The 2C feedwater heater drain valve actuator is scheduled to be replaced during the first available load drop outage. In the interim, the valve has been manually jacked open to the normal operating position.

Previous Similar Occurrences

Licensee Event Reports (LERs) 2-94-008 and 1-95-003 reported reactor power transients during which the licensed maximum power level was momentarily exceeded. However, the causes for these previous events were different, and therefore, the corrective actions from these previous events would not have prevented the event reported by this LER.

LER 2-95-010 reported a reactor scram caused by an EHC system malfunction resulting from a relay failure. The corrective actions from this previous event would not have prevented the event reported by this LER. No previous events were identified concerning reactor scrams resulting from power excursions caused by feedwater heater isolations.

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